UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406-1415

May 15, 2007

Mrs. Mary G. Korsnick
Vice President, R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, New York 14519

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT - NRC INTEGRATED INSPECTION REPORT 05000244/2007002

Dear Mrs. Korsnick:

On March 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your R. E. Ginna facility. The enclosed integrated inspection report documents the inspection results, which were discussed on April 26, 2007, with David Holm and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission’s rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two findings of very low safety significance (Green) that were also determined to be violations of NRC requirements. Additionally, one licensee-identified violation, which was determined to be of very low safety significance is listed in this report. Because the violations were of very low safety significance and were entered into your corrective action program (CAP), the NRC is treating these violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at R.E. Ginna Nuclear Power Plant.

In accordance with 10 CFR 2.390 of the NRC’s “Rules of Practice,” a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the

NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Blake D. Welling, Acting Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket No. 50-244
License No. DPR-18

Enclosure: Inspection Report 05000244/2007002
w/ Attachment: Supplemental Information

cc w/encl:
M. J. Wallace, President, Constellation Generation
J. M. Heffley, Senior Vice President and Chief Nuclear Officer
P. Eddy, Electric Division, NYS Department of Public Service
C. Donaldson, Esquire, Assistant Attorney General, New York Department of Law
C. W. Fleming, Esquire, Senior Counsel, Constellation Energy Group, Inc.
M. Balboni, New York State Deputy Secretary for Public Safety
J. Spath, Program Director, New York State Energy Research and Development Authority
T. Wideman, Director, Wayne County Emergency Management Office
M. Meisenzahl, Administrator, Monroe County, Office of Emergency Preparedness
T. Judson, Central New York Citizens Awareness Network

NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,
/RA/

Blake D. Welling, Acting Chief
Reactor Projects Branch 1
Division of Reactor Projects

Distribution w/encl.:
S. Collins, RA
M. Dapas, DRA
J. Lamb, RI OEDO
J. Lubinski, NRR
D. Pickett, PM, NRR
B. Welling (Acting Branch Chief), DRP
K. Kolaczyk, DRP, Senior Resident Inspector
M. Marshfield, DRP, Resident Inspector
K. Kolek, DRP, Resident OA
Region I Docket Room (with concurrences)
ROPreports@nrc.gov

DOCUMENT NAME: G:\DRP\BRANCH1\Ginna\Reports\2007-002\IR 2007-002 Rev. 2.wpd
SUNSI Review Complete: BDW (Reviewer's Initials)
After declaring this document "An Official Agency Record" it **will** be released to the Public.
To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure **"E"** = Copy with attachment/enclosure **"N"** = No copy

OFFICE	RI/DRP		RI/DRP		RI/DRP	
NAME	KKolaczyk		BBickett		BWelling	
DATE	05/08/07		05/08/07		05/10/07	

OFFICIAL RECORD COPY

ML071350196

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-244
License No.:	DPR-18
Report No.:	05000244/2007002
Licensee:	Constellation Energy, R. E. Ginna Nuclear Power Plant, LLC (Constellation)
Facility:	R. E. Ginna Nuclear Power Plant (Ginna)
Location:	Ontario, New York
Dates:	January 1, 2007 through March 31, 2007
Inspectors:	K. Kolaczyk, Senior Resident Inspector M. Marshfield, Resident Inspector D. Silk, Senior Operations Engineer
Approved by:	Blake D. Welling, Acting Chief Reactor Projects Branch 1 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000244/2007-002; 01/01/2007 - 03/31/2007; R. E. Ginna Nuclear Power Plant; Event Followup; Drill Evaluation.

The report covered a 3-month period of inspection by resident inspectors and announced inspections by regional specialists. Two Green findings, both of which were non-cited violations (NCV), were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

Green. A self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified because Ginna failed to control the proper design configuration of installed plant equipment. Specifically, Ginna failed to update records and procedures reflecting the design requirement for a vent hole to be drilled in the exhaust port plug for the main steam isolation valve (MSIV) air actuators. As a result, a replacement actuator was installed during the October 2006 refueling outage on the "B" MSIV with a solid vent plug. This caused an inadvertent closure of the MSIV on March 16, 2007, and resulted in a reactor trip. Ginna replaced the actuator with a modified version and placed this issue in the corrective action program.

The finding is more than minor because it is associated with the design control attribute of the Initiating Events cornerstone, and it adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability during power operations. Specifically, the closure of "B" MSIV caused a reactor trip with a safety injection system actuation. The inspectors determined the finding was of very low safety significance (Green) using a Phase 1 screening of the finding in accordance with IMC 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations." The finding screened to Green because it did not contribute to the likelihood of a primary or secondary system loss-of-coolant-accident (LOCA) initiator, or to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available. (Section 4OA3.5)

Cornerstone: Emergency Preparedness

Green. The inspectors identified an NCV of 10 CFR 50.47(b)(15), radiological emergency response training, when they noted that the assigned Emergency Response Organization (ERO) communicators have not been fully trained on all communicator responsibilities as outlined in Emergency Plan Implementing Procedure (EPIP) 5-7. For example, since December 2006, contrary to EPIP 5-7, maintenance personnel who were filling the role of ERO communicator have not been trained to respond to the control room when medical and fire events have occurred at the station and properly implement their communicator duties. Ginna issued a condition report to address the training deficiency.

The inspectors determined that the failure to ensure that control room communicators were fully trained on ERO communicator responsibilities as described in procedure EPIP 5-7 was more than minor because it was associated with the ERO readiness aspect of the Emergency Preparedness cornerstone, and it affected the objective to ensure Ginna is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The EP SDP was used to assess the safety significance of this finding related to the non-risk significant planning standard 10 CFR 50.47(b)(15). Based on IMC 0609 Appendix B, “Emergency Preparedness SDP” Sheet 1 for the failure to comply with an NRC requirement and the examples provided in Section 4.15, this finding was determined to be of very low safety significance (Green). The finding screened to Green, because the individuals were not trained to the expectations outlined in EPIP 5-7; however, they had received training on their communicator duties for declared events. This finding has a cross-cutting aspect in the area of human performance, because Ginna maintenance personnel who were filling the role of ERO communicator were not fully trained on the roles and responsibilities of the position as outlined in EPIP 5-7. (Section 1EP6)

B. Licensee-Identified Violation

A violation of very low safety significance, which was identified by Ginna, has been reviewed by the inspectors. Corrective actions taken or planned by Ginna have been entered into Ginna’s corrective action program (CAP). The violation and corrective action(s) are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

Ginna began the period at full Rated Thermal Power (RTP). The reactor tripped on January 27, 2007, because of a failure in the plant turbine control system and returned to full RTP on February 1, 2007. The reactor tripped a second time on March 16, 2007, when the “B” MSIV inadvertently went closed, and returned to full RTP on March 19, 2007. The plant operated at full RTP for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 site sample)

a. Inspection Scope

On January 26 and 29, 2007, the inspectors toured areas of the plant that contained equipment and systems that could be adversely affected by cold temperatures. Areas of focus were the intake structure, auxiliary building, the standby auxiliary feedwater (SAF) pump room, and “A” and “B” battery rooms. During the tour, the inspectors verified that temperatures in those rooms did not decrease below the values outlined in the plant Updated Final Safety Analysis Report (UFSAR). The inspectors used Ginna Procedure A-54.4.1, "Cold Weather Walkdown Procedure," as a guide during plant walkdowns.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial Walkdown (3 - samples)

a. Inspection Scope

The inspectors used plant technical specifications (TS), Ginna operating procedures, plant piping and instrument drawings (P&IDs), and the UFSAR as guidance for conducting partial system walkdowns. The inspection reviewed the alignment of system valves and electrical breakers to ensure proper in-service or standby configurations as described in plant procedures and drawings. During the walkdown, the inspectors evaluated material conditions of the system and adjacent areas. The inspectors also verified that operations personnel were following plant TS. The following plant system alignments were reviewed:

- On February 7, 2007, the containment purge system was walked down by the inspectors. This system was examined since improper system alignment could result in a loss of containment integrity.

- On February 13, 2007, the inspectors performed a walkdown of the "A" emergency diesel generator (EDG) system while the "B" EDG was unavailable for planned monthly maintenance. This system was examined because it would have been the primary backup power supply in the event of a loss of offsite power.

- On March 28, 2007, the inspectors performed a walkdown of the boric acid transfer system for reactor reactivity control. The system was walked down because of the its significance to reactor control.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown (2 - samples)

a. Inspection Scope

The inspectors performed a detailed walkdown of the alignment and condition of the containment mini-purge system. The mini-purge system was chosen because proper operation, testing and maintenance of the system ensure the containment will remain functional during certain accident scenarios. In addition to verifying proper system alignment as specified by plant TS, the plant UFSAR, and Ginna procedures and drawings, the inspector reviewed system maintenance and condition reports (CRs). None of the CRs or maintenance work orders indicated the performance or reliability of the system had declined.

The inspectors performed a detailed walkdown of the alignment and condition of the 125 volt direct current (DC) and Class 1E emergency batteries. The DC power system was chosen because of the important role it plays in accident mitigation in the event of a loss of offsite power. In addition to verifying proper system alignment as specified by plant TS, the plant UFSAR, NRC Generic Letters (GLs) and Information Notices, and Ginna procedures and drawings, the inspector reviewed system maintenance and CRs. None of the CRs or maintenance work orders indicated the performance or reliability of the system had declined.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection (11 - samples)

a. Inspection Scope

Using the Ginna Fire Protection Program documents as a guide, the inspectors performed walkdowns of the following fire areas to determine if there was adequate control of transient combustibles and ignition sources. The material condition of fire protection systems, equipment and features, and the material conditions of fire barriers were also inspected against industry standards. In addition, the passive fire protection features were inspected, including the ventilation system fire dampers, structural steel fire proofing, and electrical penetration seals. The following plant areas were inspected:

- Intermediate Building Basement, (Fire Zone IBN-1);
- Cable Tunnel, (Fire Area CT);
- Standby Auxiliary Feedwater Pump Building, (Fire Area SAF);
- Technical Support Center, South Corridor, (Fire Zone IS);
- Technical Support Center, Mechanical Equipment Room, (Fire Zone IM);
- Screenhouse Basement, (Fire Zone SH-1);
- Screenhouse Operating Floor, (Fire Zone SH-2);
- Screenhouse Circulating Water Pump Area, (Fire Zone SH-3);
- Water Treatment Room, (Fire Zone SB-1WT);
- Control Room, (Fire Area CC); and
- Relay Room, (Fire Zone RR).

b Findings

No findings of significance were identified.

.2 Fire Brigade Drill - Annual Sample (1 - sample)

a. Inspection Scope

The inspectors observed an announced test of the Ginna station fire brigade on March 9, 2007. The test involved a simulated fire in two of the four chlorine injection pumps that are located in the basement of the screenhouse building. The inspectors verified the fire brigade personnel, which consisted of five auxiliary operators, responded within the time lines outlined in the Ginna Fire Protection Program Report, and used personnel protective equipment specified by Ginna fire fighting procedures. The inspector verified the brigade used proper firefighting techniques and performed satisfactorily as a team. Following the drill, the inspectors observed the post-drill critique.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 internal sample)

a. Inspection Scope

To evaluate Ginna's internal flood protection measures, the inspectors reviewed the preventive maintenance program for the turbine building (TB) condenser pit sump pumps, and reviewed how Ginna personnel assessed the information contained in NRC Information Notices (IN) 2003–08, "Potential Flooding Through Unsealed Concrete Floor Cracks," IN 2005-11, "Internal Flooding/Spraydown of Safety-Related Equipment Due to Unsealed Equipment Hatch Floor Plugs and/or Floor Drains," and IN 2005-30, "Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design." The inspectors also reviewed the plant UFSAR and drawings of the TB floor drain systems, and walked-down the circulating water pump flood mitigating trip sensors in the screenhouse, TB, and portions of the TB floor drain and sump pump systems.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07 - 2 samples)

a. Inspection Scope

The inspectors reviewed performance tests, periodic cleaning, eddy current inspections, chemical control methods, tube leak monitoring, tube plugging condition, operating procedures and maintenance practices for the "A" and "B" closed cooling water (CCW) heat exchangers (HXs) and the "A" and "B" EDG jacket water and lube oil HXs. The purpose of the review was to verify that controls for the selected components conformed to Ginna's commitments to GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

The inspectors compared the inspection results for the heat exchangers to the established acceptance criteria to verify that the results were acceptable and that the heat exchangers operated in accordance with design. The inspectors walked down the systems, structures, and components. The inspectors reviewed system health reports and interviewed applicable system engineers.

The inspectors also reviewed a sample of CRs related to the CCW HXs, and EDG HXs, to ensure that Ginna was appropriately identifying, characterizing, and resolving problems related to these systems and components within regulatory requirements and Ginna's commitments.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (1 - sample)

a. Inspection Scope

On January 10, 2007, the inspectors observed a licensed operator simulator scenario. The test observed was scenario ES1213-14, “Ejected RCCA.” The inspectors reviewed the critical tasks associated with the scenario, observed the operators’ performance, and observed the post-evaluation critique by the evaluation group. The inspectors also reviewed and verified compliance with Ginna procedure OTG-2.2, “Simulator Examination Instructions,” and assessed the Ginna evaluators’ compliance with guidance contained in that document.

b. Findings

No findings of significance were identified.

.2 Biennial Review (1 - sample)

a. Inspection Scope

The following inspection activities were performed using NUREG-1021, Revision 9, “Operator Licensing Examination Standards for Power Reactors,” Inspection Procedure Attachment 71111.11, “Licensed Operator Requalification Program,” and NRC Manual Chapter 0609, Appendix I, “Operator Requalification Human Performance SDP,” 10 CFR 55.46 Simulator Rule (sampling basis) as acceptance criteria.

The inspectors reviewed documentation of operating history since the last requalification program inspection. The inspectors also discussed facility operating events with the resident staff. Documents reviewed included NRC inspection reports, plant performance insights, and CRs that involved human performance issues for licensed operators to ensure that operational events were not indicative of possible training deficiencies.

The inspectors reviewed two exam sets for both the comprehensive reactor operator and senior reactor operator biennial written exams, as well as scenarios and job performance measures (JPMs) administered during this current exam cycle to ensure the quality of these exams met or exceeded the criteria established in the Examination Standards and 10 CFR 55.59.

During the onsite weeks of this inspection, the inspectors observed the administration of operating examinations to operating crews “A” and “C.” The operating examinations consisted of two to three simulator scenarios for each crew and one set of five JPMs administered to each individual.

Conformance with Simulator Requirements Specified in 10 CFR 55.46

For the site specific simulator, the inspectors observed simulator performance during the conduct of the examinations, and discrepancy reports to verify compliance with the requirements of 10 CFR 55.46. The inspectors reviewed simulator maintenance, testing and control procedures and discussed simulator maintenance, testing, configuration control and machine operation with members of the simulator maintenance staff. A sample of simulator tests including transients, core performance, computer real time, and steady state were also reviewed by the inspectors. Inspectors verified that a sample of completed simulator CRs from the past two-year period effectively addressed the described issue. For a listing of the specific simulator tests reviewed see the attachment for list of documents reviewed.

Conformance with Operator License Conditions

Conformance with operator license conditions was verified by reviewing the following records:

- Remediation training records for three individuals and one crew were reviewed for the past two-year training cycle;
- Proficiency watch-standing and reactivation records. A sample of licensed operator reactivation records were reviewed as well as a random sample of watch-standing documentation for time on shift to verify currency and conformance with the requirements of 10 CFR 55; and
- Restoration to active license status of two licensed operators.

Licensee's Feedback System

The inspectors interviewed instructors, training/operations management personnel, and three operators for feedback regarding the implementation of the licensed operator requalification program to ensure the requalification program was meeting their needs and responsive to their noted deficiencies/recommended changes. The inspectors also confirmed that selected plant and industry events were incorporated into the requalification program.

On February 9, 2007, the inspectors performed an in-office review of Ginna requalification exam results. These results included the annual operating tests administered this year. The inspection assessed whether pass rates were consistent with the guidance of NRC IMC 0609, Appendix I, "Operator Requalification Human Performance SDP." The inspectors verified that:

- Crew failure rate on the dynamic simulator was less than 20 percent. (Failure rate was 14.3 percent);

- Individual failure rate on the dynamic simulator test was less than or equal to 20 percent. (Failure rate was 14.3 percent);
- Individual failure rate on the walkthrough test (JPMs) was less than or equal to 20 percent. (Failure rate was 0.0 percent);
- Individual failure rate on the comprehensive biennial written exam was less than or equal to 20 percent. (Failure rate was 5.7 percent); and
- More than 75 percent of the individuals passed all portions of the exam (85.7 percent of the individuals passed all portions of the exam).

As per Ginna's program, individuals failing any portion of the requalification examination are put through remediation training and must pass a retest before being returned to shift.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 - 2 samples)

a. Inspection Scope

The inspectors evaluated Ginna's work practices and follow-up corrective actions for selected system, structure, or component (SSC) issues to assess the effectiveness of Ginna's maintenance activities. The inspectors reviewed the performance history of those SSCs and assessed Ginna's extent-of-condition determinations for those issues with potential common cause or generic implications to evaluate the adequacy of Ginna's corrective actions. The inspectors reviewed Ginna's problem identification and resolution actions for these issues to evaluate whether Ginna had appropriately monitored, evaluated, and dispositioned the issues in accordance with Ginna procedures and the requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance." In addition, the inspectors reviewed selected SSC classification, performance criteria and goals, and Ginna's corrective actions that were taken or planned, to verify whether the actions were reasonable and appropriate. The following issues were reviewed:

- Failure of control room cooling air compressor AKR01A on November 26, 2006, because of a leaking check valve; and
- Main battery "A" vital battery monitor maintenance status.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 8 samples)

a. Inspection Scope

The inspectors evaluated the effectiveness of Ginna's maintenance risk assessments required by paragraph a(4) of 10 CFR 50.65. This inspection included discussions with control room operators and scheduling department personnel regarding the use of Ginna's online risk monitoring software. The inspectors reviewed equipment tracking documentation and daily work schedules, and performed plant tours to gain reasonable assurance that actual plant configuration matched the assessed configuration. Additionally, the inspectors verified that Ginna's risk management actions, for both planned and/or emergent work, were consistent with those described in procedure IP-PSH-2, "Integrated Work Schedule Risk Management." Risk assessments for the following out-of-service SSCs were reviewed:

- Maintenance risk associated with a secondary system transient caused when the condensate makeup valve failed and subsequent repairs to valve 4315. (January 5, 2007);
- Planned maintenance on the “A,” “B” and “C” safety injection pumps to test service water coolant flow to the pump bearings. Testing was performed utilizing a different methodology to measure flow. (January 11, 2007);
- Planned maintenance on the “A” EDG, RSSP-19, Diesel Generator “A” - Auto-Start Undervoltage Logic Test. Initiation of the test caused the equipment out-of-service monitor to indicate a higher than expected “orange” risk when initially attempted on February 8, 2007. The test was completed on March 20, 2007;
- Planned maintenance to replace pressure control valve PCV-135 (February 14, 2007);
- Unplanned maintenance when the radioactive waste system gas analyzer failed on February 20, 2007. The system was restored on February 23, 2007;
- Planned maintenance to remove corrosion susceptible elements from secondary systems found to contain brass fittings and copper piping. A failed brass fitting was responsible for a small steam leak on a level detector on the 2B main stream reheater (MSR) on February 20, 2007. Work continued for several weeks to correct all the issues identified as extent of condition associated with the initial identified steam leak (Work began the week of February 26, 2007);
- Risk associated with an identified failure to properly correct a fire breach permit in the wall separating the intermediate building and the TB. (March 5, 2007); and
- Risk associated with a new version of procedure PT 2.2Q, “Residual Heat Removal System - Quarterly,” which initially indicated a higher than expected risk value for quarterly pump testing. (March 26, 2007).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 5 samples)

a. Inspection Scope

The inspectors reviewed operability determinations to verify that the operability of systems important to safety were properly established, that the affected components or systems remained capable of performing their intended safety functions, and that no unrecognized increase in plant or public risk occurred. In addition, the inspectors reviewed the following operability evaluations to determine if system operability was properly justified in accordance with IP-CAP-1.1, "Technical Evaluation for Current Operability and Past Operability Determination Worksheet":

- 2007-000876, turbine driven auxiliary feedwater (TDAFW) pump rotating with steam supply valves closed;
- 2007-001302, TDAFW pump steam supply check valve 3505B puffing;
- 2007-000729, containment spray piping schedule does not match P&ID;
- 2007-002130, main steam non-return check valves failed closure acceptance criteria per PT-2.10.15; and
- 2007-002021, TM-402B (Delta T SP1 module) lag time found out of specification.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17 - 1 sample)

a. Inspection Scope

The inspectors reviewed plant change request (PCR) 2006-0031, "Control Room Raised Floor Upgrade" which was installed in phases during the fourth quarter 2006 and first quarter 2007. The modification updated and rearranged the workstations located in the control room, which included installing new flat monitors on the control room panels, and elevating the desks of the Shift Technical Advisor and Control Room Supervisor. New carpeting and wall treatments were also installed. The inspectors reviewed PCR 2006-0031, and compared the document to Ginna design basis information including the Ginna Fire Hazards Analysis Report. As part of the review the inspector observed portions of the modification installation.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 8 samples)

a. Inspection Scope

The inspectors observed portions of post-maintenance testing activities in the field to determine whether the tests were performed in accordance with approved procedures. The inspectors assessed the test's adequacy by comparing the test methodology to the scope of maintenance work performed. In addition, the inspectors evaluated the test acceptance criteria to verify that the tested components satisfied the applicable design and licensing bases and technical specification requirements. The inspectors reviewed the recorded test data to determine whether the acceptance criteria were satisfied. The following post-maintenance testing activities were reviewed:

- Standby auxiliary feedwater pump "D" breaker replacement retest, white light testing in work package and PT-36Q-D, "Standby Auxiliary Feedwater Pump D - Quarterly," (January 25, 2007);
- PT-12.1, EDG "A" following Fuel Oil Booster Pump Replacement (February 8, 2007);
- PT-31, Charging Pump Inservice Test, Minimum Flow Test following replacement of the "A" charging pump drive belt (February 12, 2007);
- PT-17.1, Performance Test of Area Radiation Monitors and High Range Area Radiation Monitors, following repairs to RM-10B (March 14, 2007);
- PT-60.13A, Control Room Emergency Air Treatment System (CREATS) Heating and Cooling System Performance Test - Train "A" completed after repairs to coolant system due to short cycling (March 15, 2007);
- CPI-FT-464, Calibration of Steam Generator a Steam Flow Transmitter FT-464, following damaged caused during isolation of an associated condenser pot during the forced outage March 16-18, 2007 (March 18, 2007);
- PT-2.10.5, MSIV Shutdown Exercising Requirements, conducted following replacement of the MSIV "B" valve operator when the valve failed closed causing a plant trip on March 16, 2007 (March 18, 2007); and
- PT-22.1, Equipment Hatch Door Seal Leakrate Test, conducted following repairs to the inner door (March 29, 2007).

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 2 samples, forced outage)

.1 Reactor Trip Due to Electro-Hydraulic Control System Failure

a. Inspection Scope

On January 27, 2007, the plant tripped as a result of a loss of load transient caused by a failure in the main turbine electro-hydraulic control (EHC) system. The plant response to the event was similar to previous loss of load events at Ginna, with the power-operated relief valves lifting twice to control pressure in the reactor coolant system. During the outage, the inspectors reviewed the control of plant shutdown risk and Ginna post trip followup work activities. Startup preparations and activities were observed by the inspectors. These inspection processes constituted one sample of this inspection procedure for a forced outage.

b. Findings

No findings of significance were identified.

.2 Reactor Trip Due to Main Steam Isolation Valve “B” Failing Closed

a. Inspection Scope

On March 16, 2007, the plant tripped as a result of a valid reactor trip and safety injection signal generated when the “B” MSIV went shut while operating at 100 percent power. During the outage, the inspectors reviewed the control of plant shutdown risk and the repairs to the MSIV operator. Following the trip, Ginna performed a walkdown of containment and identified several valve packing leaks and a minor swagelock steam leak on a steam flow detector, which were repaired. When maintenance activities were complete, the inspectors observed several plant startup activities. These inspection processes constituted one sample of this inspection procedure for a forced outage.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors witnessed the performance and/or reviewed test data for the following six surveillance tests that are associated with selected risk-significant SSCs to verify that TS were followed, and that acceptance criteria were properly specified. The inspectors also verified that proper test conditions were established as specified in the procedures, and that no equipment preconditioning activities occurred, and that acceptance criteria had been met.

- PT-16Q-B, Auxiliary Feedwater Pump B Quarterly, (January 4, 2007);
- PT-12.2, EDG B (January 17, 2007);
- PT-32B, Reactor Trip Breaker Testing - Train B (January 19, 2007);
- PT-12.3, Security Emergency Diesel Test (February 7, 2007);
- PT-2.2Q, Residual Heat Removal System - Quarterly (“B” Pump Only) (February 9, 2007); and
- PT-13, Fire Pump Operation and System Alignment (March 30, 2007).

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors reviewed the following temporary plant modification to determine whether the temporary change adversely affected system or support system availability, or adversely affected a function important to plant safety. The inspectors reviewed the associated system design bases, including the UFSAR and TS, and assessed the adequacy of the safety determination screening and evaluation. The inspectors also assessed configuration control of the temporary change by reviewing selected drawings and procedures to verify whether appropriate updates had been made. The inspectors compared the actual installation with the temporary modification documents to determine whether the implemented change was consistent with the approved documented modification. The inspectors reviewed the post-installation test results to verify whether the actual impact of the temporary change had been adequately demonstrated by the test. The temporary modification was reviewed by the inspectors to verify it was installed in conformance with the instructions contained in procedure IP-DES-3, “Temporary Modifications.”

- 2005-0021 Temporary leak repair to the floor drain pipe leading from the Intermediate Building basement

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

On January 10, 2007, the inspectors observed scenario ES1213-14, “Ejected RCCA,” a licensed operator simulator scenario that included a limited test of the Ginna emergency response plan. During the exercise, the crew successfully classified the event in a timely manner, and the drill was counted as a success in the Ginna “Drill/Exercise Performance” performance indicator.

The inspectors also reviewed issues related to the training, roles and responsibilities of Emergency Response Organization communicators.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR 50.47(b)(15), which requires that radiological emergency response training be provided to those who may be called on to assist in an emergency. Specifically, the station's control room communicators have not been fully trained on all ERO communicator responsibilities.

Description: EPIP 5-7, "Emergency Organization," describes the roles and responsibilities of personnel assigned to the ERO during a declared emergency event, i.e., Unusual Event or higher at Ginna. One of the individuals in the ERO is the control room communicator. EPIP 5-7 states that the control room communicator is required to report to the control room and make the Shift Manager aware of their presence at the announcement of an emergency (fire, medical, radiation, etc.). Among other duties, the communicator is tasked with notifying offsite organizations that an event has occurred at Ginna and coordinating the arrival and movement of emergency response equipment that has arrived on site to address the casualty. To perform these tasks, the communicator must proficiently operate specialized communication equipment in the control room to ensure the offsite agencies are notified in the timely manner. They must be knowledgeable of how to coordinate the arrival of offsite fire equipment and ambulances on site. Since December 2006, the control room communicator position has been staffed by individuals from the maintenance department. Prior to December 2006, the communicator position was staffed by auxiliary operators.

Since maintenance personnel have assumed the role of communicator, the inspectors have noted that maintenance personnel have not been trained or provided guidance on certain aspects of the communicator position. For example, the inspectors observed that they have not routinely reported to the control room when medical emergencies and fire drills have occurred. During these events, "extra" control room personnel that are not listed in the control room logs as the site communicator, have filled the role of communicator rather than the designated individual. During a March 9, 2007, fire drill, although the communicator reported to the control room to participate in the fire drill, the individual reported only after being summoned by control room personnel. The inspectors also observed that although the communicators are responsible for coordinating the arrival of offsite response equipment, they had not received training on how to perform this task. The difference between the requirements of EPIP 5-7 and what training the communicators had received was documented in CR 2007-001975, "Verify Maintenance Personnel are Trained and Proficient in Responsibilities Delineated in EPIP 5-7."

The inspectors noted that by not routinely participating in drills, the communicators may lose their proficiency because the equipment that is used in declared ERO events is identical to the equipment used in fires and medical emergencies. During a subsequent fire drill, in which the communicator participated, the individual was not able to locate the procedures and equipment needed to notify offsite organizations during a simulated event. This issue was documented in CR 2007-002095, "Weakness in Executing Communicator Duties."

Analysis: The inspectors determined that the failure to ensure that control room communicators were fully trained on ERO communicator responsibilities as described in procedure EPIP 5-7 was more than minor because it was associated with the ERO readiness aspect of the Emergency Preparedness cornerstone, and it affected the objective to ensure Ginna is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The EP SDP was used to assess the safety significance of this finding related to the non-risk significant planning standard 10 CFR 50. 47(b)(15). Based on IMC 0609 Appendix B, "Emergency Preparedness SDP" Sheet 1 for the failure to comply with an NRC requirement and the examples provided in Section 4.15, this finding was determined to be of very low safety significance (Green). The finding screened to Green, because the individuals were not fully trained to the requirements outlined in EPIP 5-7; however, they had received training on their communicator duties for declared events. This finding has a cross-cutting aspect in the area of human performance because Ginna maintenance personnel who were fulfilling the role of ERO communicator were not properly trained on the roles and responsibilities of the position as outlined in EPIP 5-7.

Enforcement: 10 CFR 50.47(b)(15) requires that radiological emergency response training be provided to those who may be called on to assist in an emergency. Section IV of 10 CFR 50, Appendix E, states, in part, that this training shall include emergency personnel who are involved in fire control and first aid rescue teams. EPIP 5-7, "Emergency Organization" states that the ERO communicator is tasked with notifying offsite organizations that an event has occurred at Ginna and coordinating the arrival and movement of emergency response equipment that has arrived on site to address the casualty. Contrary to 10 CFR 50.47(b)(15), since December 2006, ERO communicators have not received training that is required by Section IV of 10 CFR 50, Appendix E, so they can meet their duties as outlined in EPIP 5-7. As a result, since December 2006, maintenance personnel who were filling the role of ERO communicator have not routinely responded to the control room when medical and fire events have occurred at the station, and one occasion during a fire drill, the ERO communicator was not able to implement their portion of the Ginna emergency response plan. Because this violation was determined to be of very low safety significance and Ginna entered the deficiency into their corrective action system in CR 2007-0001975 it is being treated as a NCV, consistent with section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000244/2007002-01, Ginna Communicators Not Adequately Trained To Implement EPIP 5-7)**

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 2 samples)

a. Inspection Scope

Using the criteria specified in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 4, the inspectors verified the completeness and accuracy of the performance indicator data for unplanned scrams per 7,000 critical hours and scrams with loss of normal heat removal. To verify the accuracy of the data, the inspector reviewed monthly operating reports, NRC inspection reports and Ginna event reports issued during calendar year 2006.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

.1 Continuous Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems," and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the Ginna CAP. This review was accomplished by reviewing paper copies of CRs, attending daily screening meetings, and accessing Ginna's computerized database.

b. Findings

No findings of significance were identified.

.2 Annual PI&R Sample - NUS Modules (71152 - 1 sample)

a. Inspection Scope

Following startup from the 2006 refuel outage, Ginna personnel identified that two signal conditioning modules in the reactor protection overpower differential temperature trip circuitry were not operating properly. The modules were manufactured by NUS corporation, and were installed during the refuel outage as part of the extended power uprate project. An examination of the modules by Ginna and NUS personnel determined that the failures were caused by abraided wires which had shorted to ground. The wires were apparently damaged during manufacture of the modules. These issues were documented in CR-2006-06132, and Ginna's corrective actions consisted of taping the defective wires and examining the remaining NUS modules that were installed during the refuel outage. On December 14, 2006, NUS Corporation informed the NRC of the manufacturing defect per 10 CFR 21. The inspectors reviewed the corrective actions outlined in CR 2006-06132 and several other CRs relating to the reliability of the NUS modules against the requirements of the CAP to ensure that the full extent of the issues were identified and that the proposed corrective actions are appropriate. The inspectors examined two of the defective modules and interviewed relevant station personnel.

b. Findings and Observations

No findings of significance were identified. The corrective action in progress to examine similar NUS modules installed in the plant and in stock is appropriate. The corrective action is being given prioritization and oversight commensurate to its safety and risk significance.

4OA3 Event Followup (71153)

.1 Inadvertent Deluge System Initiation

a. Inspection Scope (1 - sample)

On January 11, 2007, Ginna operations personnel conducting a performance test on the control building air handling room deluge system, SO6, caused fire suppression panels S22 and S23 to be wetted by backspray from the discharge of the SO6 system. Wetted panel S22, which controls the deluge systems for 12A off-site power supply transformer initiated a deluge of the 12A transformer. The inspectors followed Ginna's response to the event, the corrective actions taken and the compensatory actions put in place while the wetted panels were removed from service. There was no impact on the plant or the transformers by the wetting of the S22 and S23 deluge systems. The fire suppression panels were restored to service on January 13, 2007.

b. Findings

No findings of significance were identified.

.2 Reactor Trip Due to Turbine Control System Malfunction

a. Inspection Scope (1 - sample)

On January 27, 2007, the Ginna reactor tripped from 100 percent power as a result of a valid over temperature differential temperature signal that was created when all four turbine control valves closed due to a faulty card in the turbine EHC system. Plant systems and components responded as designed with both power operated relief valves, and one steam generator safety valve lifting to relieve excess primary and secondary plant pressure. Following the reactor plant trip, a minor chemistry transient in the secondary plant occurred when circulating water entered the feed and condensate system through a failed check valve in the steam generator blowdown system. This event was similar to an event that had occurred in July 2005. Following the trip, the plant was stabilized in Mode 3, "Hot Shutdown," with steam generator level and pressure being controlled by the steam-driven auxiliary feedwater pump and atmospheric relief valve systems.

The resident inspectors reviewed Ginna’s post-trip assessment, and plant restart activities. An independent post-trip review was performed by the inspectors.

b. Findings

No findings of significance were identified.

.3 Steam Leak on Main Steam Reheater (MSR)

a. Inspection Scope (1 - sample)

On February 20, 2007, a brass fitting on the MSR 2B failed, resulting in a steam leak heard by operators in the turbine building. The leak was isolated by operators by closing two valves locally in the vicinity of the failed component. A subsequent evaluation of the failed fitting indicated that the part had failed due to excessive corrosion of the brass fitting in the high pH environment of the secondary systems. The acceptability of the brass fittings had been questioned previously by plant personnel in January of 2006, but were not replaced when engineering personnel determined that pressure rating of the fitting was acceptable. Although the pressure rating was acceptable, the ability of the fitting to remain operable under the corrosive affects of the higher pH environment of the secondary plant systems was not considered. The secondary pH was raised to reduce iron transport to the new steam generators after their installation in 1996. The failed fitting was likely installed during a major refit of all the MSR components in 1984. Ginna has subsequently identified and replaced several brass fittings since the event as part of their extent of condition response to this event.

b. Findings

No findings of significance were identified.

.4 (Closed) LER 05000244/2006007-01 Main Steam Safety Valve (MSSV) Setpoint Exceedance

This LER is an update of an event evaluated in the 4th quarter 2006 NRC inspection report 05000244/2006005. That report generated an unresolved item, which was opened pending Ginna's determination of the root cause(s) for the MSSV failures, and identification of a possible performance deficiency which may have caused two of eight main steam relief valves to lift at a pressure above the acceptance band during testing in October 2006. The URI is closed in section 4OA5 of this report. The apparent cause for the failure of the valves to lift prior to exceeding the limiting pressure was determined to be increased friction in the spindle guide area of the valve. Ginna is evaluating whether to refurbish the valves during the next refueling outage. The enforcement aspects of this issue are discussed in Section 4OA7 of this report. This LER is closed.

.5 Reactor Trip Due to Inadvertent MSIV Closure

a. Inspection Scope (1 - sample)

On March 16, 2007, the station experienced a reactor trip from 100 percent power as a result of the “B” MSIV failing shut and causing a valid safety injection signal on low steam pressure in the “A” steam flow header, with an associated trip and subsequent MSIV isolation of the “A” MSIV on high steam flow in the same header. Plant systems and components responded as designed. No primary power-operated relief valves lifted during the event. At least one steam generator safety valve on the “B” steam generator lifted to relieve excess pressure in the isolated “B” steam generator. Following the trip, the plant was stabilized in Mode 3, "Hot Shutdown," with steam generator level and pressure being controlled by the motor auxiliary feedwater pumps and atmospheric relief valves.

The resident inspectors reviewed Ginna’s post-trip assessment, and plant restart activities. An independent post-trip review was performed by the inspectors.

b. Findings

Introduction: A self-revealing NCV of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” was identified because Ginna failed to control the proper design configuration of installed plant equipment. Specifically, during the October 2006 refuel outage, an improperly designed air actuator was installed on the “B” MSIV. The MSIV actuator subsequently failed and caused a reactor trip.

Description: Each MSIV at Ginna contains an air actuator that supplies the motive force to open and close the valves. The actuator has two main parts: a spring that holds the valve closed, and a piston assembly that is used to open the valve. Valve opening is accomplished when sufficient instrument air has been supplied to the top of the piston to overcome the force of the closing spring.

During the early 1970s, several reactor plant trips occurred at Ginna when air leakage past the actuator piston seals equalized pressure across the piston allowing the closure spring to overcome the piston opening force. To correct this condition, a vented plug was installed in the exhaust side of the actuator in place of a previously installed solid plug. However, this modification was not identified in plant design drawings, station procedures or warehouse purchase orders. This oversight did not have immediate adverse consequences, because Ginna rebuilt the MSIV actuators during planned maintenance activities, and reused the vented plugs. However, during the October 2006 refuel outage, Ginna changed the previous maintenance practice of rebuilding the installed operator, and instead installed a replacement item in the “B” MSIV which had not been modified in accordance with the vendor’s directions. As a result, similar to what occurred in the early 1970's, once sufficient air leakage past the piston seals had occurred, the “B” MSIV closed. Ginna replaced the MSIV vent plug and verified “A” MSIV was not subject to the same design error.

Analysis: The performance deficiency is a failure of Ginna to properly control the configuration of installed plant equipment, including translation of design information into plant procedures and work control documents so they reflect the desired plant configuration. The finding is more than minor because the deficiency is associated with the design control attribute of the Initiating Events cornerstone, and adversely affected the cornerstone objective of limiting the likelihood of those events that upset plant stability during power operations. Specifically, the inadvertent closure of “B” MSIV resulted in a reactor trip with safety injection system actuation. The finding was determined to be of very low safety significance (Green) in accordance with IMC 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” based on a Phase 1 analysis because the finding did not contribute to the likelihood of a primary or secondary system LOCA initiator, or to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions would not be available.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, the “B” MSIV actuator installed during the outage of October 2006 was not correctly designed because proper design information was not translated into onsite procedures and drawings associated with this modification. The MSIV operator subsequently failed shut causing a reactor trip with safety injection on March 16, 2007. Because this deficiency was determined to be of very low safety significance and Ginna entered the deficiency into their corrective action system in CR 2007-002172, it is being treated as a NCV, consistent with section VI.A.1 of the NRC enforcement policy. **(NCV 05000244/2007002-02, Failure of “B” MSIV Due to Inadequate Design Control)**

.6 (Closed) LER 05000244/2007001 Loss of Electrical Generation Results in Plant Trip

The circumstances involving this incident were previously reviewed in Section 1R20 and 4OA3.2 of this report. This LER is closed.

4OA5 Other Activities

(Closed) URI 05000244/2006005-02, Determination of Performance Deficiency Associated with TS 3.7.3, Main Steam Safety Valves

a. Inspection Scope

This item was opened pending determination of a performance deficiency, which caused two main steam relief valves to lift outside of their acceptance band during the October 2006 refuel outage. As outlined in section 4OA3.4 of this report, Ginna determined that the failure of the valves to lift within the acceptable band was caused by increased friction in the spindle guide area likely caused by small amounts of grit build up on the spindle and a manufacturer-identified tendency for bearing material to close up on the spindle over time. Ginna is evaluating whether to refurbish the valves during the next refueling outage. This is a licensee-identified finding, and the enforcement aspects of this violation are discussed in Section 4OA7 of this report. This URI is closed.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 26, 2007, the resident inspectors presented the inspection results to Mr. David Holm and other members of his staff, who acknowledged the findings. The inspectors asked Ginna's supervision whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by Ginna and is a violation of NRC requirements, which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a NCV.

TS 3.7.1 requires that all MSSVs shall be operable. Contrary to this requirement, on October 7, 2006, with the plant in Mode 1, Ginna performed in-place testing of all MSSVs and determined that two of the valves lifted outside the acceptance band. Since the two unsatisfactory as-found lift pressures may have arisen over a period of time (found during sequential testing), Ginna determined that at least one MSSV was inoperable for longer than the allowed outage time during the previous operating cycle. This condition was documented by Ginna in CR 2006-004751 and subsequently corrected. This finding is of very low safety significance because it did not increase the probability or consequences of a core damage event.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

M. Korsnick	Vice President, Ginna
D. Holm	Plant Manager
J. Yoe	Operations Manager
D. Blankenship	Manager, Radiation Protection
E. Groh	Assistant Operations Manager (Shift)
S. Kennedy	Emergency Preparedness Manager
J. Pacher	Manager, Nuclear Engineering Services
B. Randall	Nuclear Safety and Licensing Manager
W. Thomson	Chemistry Supervisor
R. Whalen	Manager, Ginna Maintenance

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000244/2007002-01	NCV	Ginna Communicators Not Adequately Trained To Implement EPIP 5-7
05000244/2007002-02	NCV	Failure of "B" MSIV due to Inadequate Design Control

Closed

05000244/2006005-02	URI	Determination of Performance Deficiency Associated with TS 3.7.3, Main Steam Safety Valves
05000244/2006007-01	LER	Main Steam Safety Valve Setpoint Exceedance
05000244/2007001-00	LER	Loss of Electrical Generation Results in Plant Trip

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

ER-SC.3, Low Screenhouse Water Level

Section 1R04: Equipment Alignment

Procedures
O-11, Control of Mini-Purge Exhaust Valves While Depressurizing Containment
PT-2.5.1, Air Operated Valves, Quarterly Surveillance Containment
PT-17.2, Area Radiation Monitors R-11 to R-22 and Iodine Monitors R-10 and R-10B
S-23.2 , Containment Purge Procedure

S-23.2.3, Containment Mini-Purge System Operation

Programs
Ginna IST Program Document
Ginna ISI Program Document

Drawings

33013-1865	Containment HVAC Systems Purge Supply
33013-1866	Containment HVAC Systems Purge Supply
33013-1870	Auxiliary/Intermediate Building HVAC Systems
03202-0102	125 VDC Power Distribution System
33013-1239	(Sheet 1 of 2) Diesel Generator - A
33013-1266	Chemical and Volume Control System Boric Acid (CVCS)

Documents

NRC Generic Letter (GL) 91-06	Resolution of Generic Issue A-30, April 29, 1991
Ginna Response to GL 91-06	Resolution of Generic Issue A-30, October 28,1991
UFSAR Section 8.3.2	DC Power Systems
NRC Information Notice 94-80	Inadequate DC Ground Detection in DC Distribution Systems
UFSAR Section 9.3.4.3.3	Reactor Makeup Control System
UFSAR Section 3A.4.2.1	Boration System

Condition Reports
2006-002581
2006-006510
2006-006978
2006-007038

Section 1R05: Fire Protection

Procedures
FRP-30.0, SH Basement
SC-3.2.11, Immediate Action - Screenhouse Fire
SC-3.3.1, Immediate Fire Notification
SC-3.4.1, Fire Brigade Captain and Control Room Personnel Responsibilities

Condition Reports
2007-000396
2007-000397

Documents
Ginna Station Fire Protection Program

Section 1R06: Flood Protection Measures

Drawings

33013-2681	Sump Pumps Drains and Sewage Pumps
33013-1912	Condensate Demineralizer Regeneration Waste Handling

Procedures
M95, Annual Inspection Maintenance and Operational Check of the Backflow Protection System

Condition Reports
2005-6560
2007-1070

Section 1R07: Heat Exchanger Performance

Documents
DA-ME-97-016, CCW and RHR HX Performance Evaluation, Rev. 0, 10/9/98
DA-ME-98-138, EDG Lube Oil and Jacket Water HX Plugging Limits and Thermal Performance at Limiting SW Flows, Rev. 1, 10/29/98
DA-ME-98-139, EDG Lube Oil and Jacket Water HX SW D/P Limits, Rev. 1, 7/9/99
DA-ME-99-067, SAFW Pump Room Coolers Thermal Performance Evaluation, Rev. 0, 4/12/00
DA-ME-2003-039, CCS HX A & B Thermal Performance Testing, performed 9/16/03, Rev. 0, 3/9/04
SWSROP, Service Water System Reliability Optimization Program, Rev. 7, 4/26/06
Repetitive Task Number P301717, EAC01A - Clean, Inspect, Eddy Current tube side [CCW HX], 4/19/06
Repetitive Task Number P301709, CMP-10-03-ESW08A/ESW09A - Clean Inspect, Eddy Current tube side [EDG HXs], 3/3/06
Repetitive Task Number P401084, Open Inspect Clean SAFW Pump Room Cooler per M-11.34.12 M-37.130 M-93, 3/10/06
CMP-10-04-EAC01A, Maintenance for EAC01A [CCW HX] performed 12/13/04, Rev. 3, 12/15/97
CMP-10-03-ESW08A, Maintenance for ESW08A [EDG JW HX] performed 10/21/05, Rev. 4, 2/23/05
CMP-10-03-ESW09A, Maintenance for ESW09A [EDG LO HX] performed 10/21/05, Rev. 5, 3/8/05

Section 1R11: Licensed Operator Requalification

Requalification Program Procedures/Documents:
OTG-2.0, Annual Examination Instructions, Revision19
OTG-2.1, Oral/Walkthrough Examination Instructions, Revision 8
OTG-2.2, Simulator Examination Instructions, Revision 39
OTG-2.3, Job Performance Measure Instructions, Revision 16
OTG-2.4, Written Examination Instructions, Revision 14
OTG-2.5, Exam Failure Review Process, Revision13
OTG-2.6, Dynamic Simulator Examination Scenarios, Revision 15
OTG-2.8, NRC Exam Security, Revision 13
OTG-10.0, License Activation, Revision3
TR-C.5.2, Licensed Operator Requalification Program, Revision 31
IP-TQS-3, Operator and Fire Brigade Physicals, Revision 5
NTG-5, Exam Trouble Card System, Revision 4
OPS-SHIFT ORG, Shift Organization, Revision 7
OPS-SHIFT-SCHEDULE, Shift Scheduling and Watch Standing Requirements, Revision 4
2005/2006 Ginna LORT Exam Sample Plan
Training Work Request 05-0483

Simulator-Related Documentation
Simulator Test Summaries for 2005 Model Upgrade Factory Acceptance Test, Site Acceptance Test and 2006 Extended Power Uprate
2006 Operating Limits Monitoring Test 14.4.1, Revision 10
2006 100% Steady State Accuracy Test 14.4.3.1, Revision 13
2006 NSSS - BOP Energy Balance Test 14.4.4.1
List of Open Simulator Deficiency Reports
List of Simulator Deficiency Reports Closed 1/1/05 thru 12/31/06
2006 Sim Core Perf Tests 14.4.6.1 Revision 13 and 14.4.6.5 Revision 0
2006 Sim Malf Test 14.4.7.3.2, "CND-2, Main Condenser Tube Leak," Revision 8
2006 Sim Malf Test 14.4.7.4.2, "CVC-2, Letdown Line Leak Outside Cntmt," Revision 8
2006 Sim Malf Test 14.4.7.5.7, "EDS-07, Loss of Instrument Bus Supply," Revision 9
SDR-2004-130, Evaluate Concern - BE06 Response
SDR-2006-014, PPCS Response to Bad Instrument Input
SDR-2006-021, Stm Line Break Inside Cntmt Does Not Have Restricted Flow
SDR-2006-026, Loss of Condensate Pumps Following a Trip
SDR-2006-032, TDAFW Unexpected Trip During ECA-0.0
SDR-2006-037, Feedwater Pumps Trip Unexpectedly During ATWS Actions

Condition Reports

CR-2007-0722	CR-2007-0721	CR-2007-0720	CR-2007-0716
CR-2007-0594	CR-2007-0532	CR-2006-7178	CR-2006-7177
CR-2006-7087	CR-2006-6534	CR-2006-6458	CR-2006-6408
CR-2006-3637	CR-2006-3939	CR-2006-3395	CR-2006-3336
CR-2006-3168	CR-2006-3251	CR-2006-2205	CR-2006-1662
CR-2006-1500	CR-2006-1453	CR-2006-1364	CR-2006-0535
CR-2006-0467	CR-2006-2072	CR-2005-6663	CR-2005-2085
CR-2005-0756	CR-2006-0067		

Biennial Written Exams 2007
Exams for Weeks One and Three

Reviewed Scenarios and JPMs - 2007 Annual Operating Exams
Exams for Weeks Two, Three, and Five

Section 1R12: Maintenance Rule Implementation

Drawings
33013-1867 Control Room Emergency Air Treatment System Cooling System

Condition Report
2007-000115

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

Procedures
PT-2.1S, A, B & C Safety Injection Pump Service Water Cooler Discharge Flow Check
RSSP-19, Diesel Generator "A" - Auto-Start Undervoltage Logic Test
S-3.2E , Placing in or Removing from Service Normal Letdown/Excess Letdown
AP-FW.1, Abnormal Main Feed Water Pump Flow or NPSH

Work Orders

20605735	Replace HC-135A, Letdown Pressure Hand Controller

Condition Reports

2007-000146	2007-001488	2007-001499	2007-001500
2007-001554	2007-001784	2007-002467	2007-002474

Documents
P&ID 33013-2280 Gas Analyzer Skid
Updated Final Safety Analysis Report (UFSAR), Section 11.3.2.2.4, Gas Analyzer
UFSAR, Section 11.5.1, Process and Effluent Radiation Monitoring and Sampling Systems Design Bases
CHA-EGSTRM Explosive Gas and Storage Tank Radioactivity Monitoring Program
EWR 4221, Revision 1, Safety Analysis Ginna Station O_2/H_2 Analyzer Replacement

Section 1R15: Operability Evaluations

Condition Reports
2007-000883
2007-000876
2007-000729
2007-002130
2007-002021 .

Documents
February 10, 1977 Letter From RG&E to the NRC
June 8, 1977 Letter From RG&E to the NRC
Licensee Event Report 76-24

Section 1R17: Permanent Plant Modifications

Documents

PCR 2006-0031	Control Room and Simulator Furniture Upgrade

Condition Reports
2007-005129
2007-001540

Section 1R19: Post Maintenance Testing

Procedures
PT-36Q-D, Standby Auxiliary Feedwater Pump D - Quarterly
PT-12.1, Emergency Diesel Generator “A”
PT-31 , Charging Pump Inservice Test
CPI-FT-464, Calibration of Steam Generator A Steam Flow Transmitter FT-464
PT-60.13A, Control Room Emergency Air Treatment System Heating and Cooling System Performance Test - Train “A”
PT-17.1, Performance Test of Area Radiation Monitors and High Range Area Radiation Monitors
PT-2.10.5, MSIV Shutdown Exercising Requirements
PT-22.1, Equipment Hatch Door Seal Leakrate Test

Work Orders

20504819	Perform PM Inspection on 52/SAFWP1D, Standby Auxiliary Feedwater Pump 1D Breaker
20700963	"A" Charging Pump Belt Disintegrated During Operation. Replace Belt
20500843	Replace the "A" Diesel Generator Fuel Oil Booster Pump as per GMM-15-01-KDG01A/B Procedure
20701655	Swagelock Fitting on top of condensing Pot is Leaking. FT-464 Damaged When Root Valves Closed Prior to Transmitter Isolation

Documents
CME-50-02-52/SAFWPD, Rev 4, Westinghouse 480V Air Circuit Breaker Type DB-50 Standby Auxiliary Feedwater Pump 'D' Breaker
UFSAR 3.1.1.5.4, Reactivity Hold Down Capability
UFSAR 3.1.2.4.4, General Design Criterion 33 - Reactor Coolant Makeup
Prompt Investigation Report for CR 2007-002566

Condition Reports
2007-001224
2007-002566

Section 1R20: Refueling and Outage Activities

Procedures
PT-20, Infrared Thermography on Secondary Side Relief Valves

Documents
Ginna Technical Evaluation of March 16, 2007 Reactor Trip
Prompt Investigation Report for CR 2007-002129, "B" MSIV Failed closed at 100% Power

Section 1R22: Surveillance Testing

Procedures
PT-12.2, Emergency Diesel Generator B
PT-32, Reactor Trip Breaker Testing - Train B
PT-12.3, Security Emergency Diesel Test
PT-2.2Q, Residual Heat Removal System - Quarterly

Section 1R23: Temporary Plant Modifications

Procedures
IP-DES-3, Temporary Modifications

Section 4OA1: Performance Indicator Verification

Documents
NEI 99-02, Nuclear Assessment Performance Indicator Guideline, Revision 4

Section 4OA2: Identification and Resolution of Problems

Procedures
—71.2 Component Rework/Test and Relay Inspection Procedure

Condition Reports

2006-006132	2006-006749	2006-007141	2007-00103
2007-000243	2007-00250	2007-001129	2007-001130
2007-001668			

Section 4OA3: Event Follow-up

Documents

PT-13.4.19 Flood Valve Testing - Suppression System SO6 Air Handling Room Auto Deluge

A-25.4 Post-Trip Review Report

LER 05000244/2006007-01 Main Steam Safety Valve Setpoint Exceedance

Condition Reports

2007-000283	2007-000290	2007-00103	2007-002126
2007-002129	2007-002134	2007-002161	2007-002172

LIST OF ACRONYMS

ADAMS	Agency-Wide Documents Access and Management System
CAP	corrective action program
CCW	closed cooling water
CFR	Code of Federal Regulation
CR	condition report
CREATS	control room emergency air treatment system
CVCS	chemical and volume control system
DC	direct current
EDG	emergency diesel generator
EHC	electro-hydraulic control
EPIP	emergency plan implementing procedure
ERO	emergency response organization
GL	generic letter
HX	heat exchanger
IMC	inspection manual chapter
IN	information notices
IST	in-service testing
JPMs	job performance measures
LOCA	loss-of-coolant accident
MSIV	main steam isolation valve
MSR	main steam reheater
MSSV	main steam safety valve
NCV	non-cited violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records
PCR	plant change request
P&ID	piping and instrument drawings
PI&R	problem, identification and resolution
PSIG	pounds per square inch
RCCA	rod cluster control assembly
RTP	rated thermal power

SAF	standby auxiliary feedwater
SDP	significance determination process
SH	screenhouse
SSC	systems, structures and components
TB	turbine building
TDAFW	turbine driven auxiliary feedwater
TS	technical specifications
UFSAR	Updated Final Safety Analysis Report